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Analysis of Pressurized Water Reactor Primary Coolant Leak Events Caused by Thermal Fatigue

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ABSTRACT: We present statistical analyses of pressurized water reactor (PWR) primary coolant leak events caused by thermal fatigue, and discuss their safety significance. Our worldwide data contain 13 leak events (through-wall cracking) in 3509 reactor-years, all in stainless steel piping with diameter less than 25 cm. Several types of data analysis show that the frequency of leak events (events per reactor-year) is increasing with plant age, and the increase is statistically significant. When an exponential trend model is assumed, the leak frequency is estimated to double every 8 years of reactor age, although this result should not be extrapolated to plants much older than 25 years. Difficulties in arresting this increase include lack of quantitative understanding of the phenomena causing thermal fatigue, lack of understanding of crack growth, and difficulty in detecting existing cracks.

1 DATA

There have been thirteen through-wall leaks caused by thermal fatigue in PWR primary coolant system piping. All occurred in piping of diameter 5–25 cm (2–10 in). Jungclaus et al. (1998) and Shah et al. (1998, 1999) document the events. They are summarized briefly in Table 1.

The World List of Nuclear Power Plants, as of December 31, 1997, is given in *Nuclear News* (1998). The PWR plants considered here include all the western-designed PWRs, mainly from the USA, western Europe, and east Asia, and the two Loviissa plants in Finland. The VVER reactors in the former German Democratic Republic were not counted. Both operating and presently shutdown reactors were counted, 217 reactors in all. For the reactors that are now shutdown, the years when the reactors were in operation were included in the analysis. For operating reactors, all years through May 31, 1998 were counted.

The dates of initial criticality were taken from the World List, except for Civaux 1, whose initial criticality was corrected (Lannoy 1998) from the preliminary date in the World List. For each thermal-fatigue leak event, the age of the plant at the time of the event was calculated. The events (plant, month, and year) are listed by plant age in the third column of Table 1.

Table 1. PWR thermal-fatigue leak data, by plant age.

Age (years)	Reactor-years at this age	Leak events caused by thermal fatigue
0.0 – 1.0	215.2	Civaux 1 5/98
1.0 – 2.0	211.6	
2.0 – 3.0	210.2	
3.0 – 4.0	208.6	
4.0 – 5.0	206.5	Crystal River 3 1/82
5.0 – 6.0	202.3	
6.0 – 7.0	200.4	Farley 2 12/87
7.0 – 8.0	198.0	
8.0 – 9.0	193.6	
9.0 – 10.0	188.3	
10.0 – 11.0	178.1	
11.0 – 12.0	167.3	Dampierre 2 9/92
12.0 – 13.0	153.5	
13.0 – 14.0	136.3	Tihange 1 6/88; Genkai 1 6/88; Loviisa 2 5/94
14.0 – 15.0	121.8	
15.0 – 16.0	110.0	
16.0 – 17.0	100.9	Dampierre 1 12/96; Loviisa 2 1/97
17.0 – 18.0	86.4	Obrigheim /86
18.0 – 19.0	74.7	Biblis-B 2/95
19.0 – 20.0	70.3	
20.0 – 21.0	63.4	
21.0 – 22.0	55.3	Three Mile Island 1 9/95
22.0 – 23.0	47.3	
23.0 – 24.0	39.3	Oconee 2 4/97
24.0 – 25.0	25.5	
25.0 – 26.0	16.3	
26.0 – 27.0	11.2	
27.0 – 28.0	8.2	
28.0 – 29.0	5.4	
29.0 – 30.0	2.7	
30.0 – 31.0	0.2	

To count the reactor-years for each age, the number of reactors that experienced 1 year, 2 years, etc. were totaled. For example, Three Mile Island 2 had its initial criticality in December 1978, and was shut down in March 1979. It was counted as 1/4 of a reactor-year (= 3 months) for reactors in year 1 of life (age 0.0 – 1.0). At the other extreme, Yankee Rowe operated from July 1961 to September 1991. It was counted as contributing 1 reactor-year at every age through 30, and 1/6 of a reactor-year for age 30.0 – 31.0. These counts of reactor-years are totaled in the second column of Table 1.

Note that the first decade of plant life contributes over 2000 reactor-years, but only 3 leaks. The second decade of plant life includes fewer reactor-years (about 1200), but more than twice as many leaks, a total of 8. This suggests a possible effect of plant age. The following analysis estimated this effect.

The rows of Table 1 were combined into 5-year bins, corresponding to ages 0.0–5.0, 5.0–10.0, etc. The reactor-years and the number of leak events were totaled for each bin. This condensation of Table 1 is shown in Table 2. The zero leak count for bin 25.0–30.0 is not surprising, because very few plants have reached that age — that age range corresponds to only 43.9 reactor-years. When a trend is estimated, as described below, the extrapolated value of leak frequency for bin 25.0–30.0 is 1.5E–2 events per reactor-year. For this frequency, it is more likely to see zero leak events in 43.9 reactor-years than to see one or more events in that time period.

Table 2. Summarized PWR thermal-fatigue leak data.

Age (years)	Reactor-years	No. of leak events	Leak events/ reactor-years
0.0 – 5.0	1052.1	2	1.9E–3
5.0 – 10.0	982.5	1	1.0E–3
10.0 – 15.0	756.9	4	5.3E–3
15.0 – 20.0	442.4	4	9.0E–3
20.0 – 25.0	230.9	2	8.7E–3
25.0 – 30.0	43.9	0	0

2 STATISTICAL ANALYSES

2.1 *Loglinear model based on event counts*

To estimate the trend, we used the data from Table 2, and assumed that the number of leaks for reactors in any 5-year age range was Poisson distributed. We also assumed that the Poisson intensity (leak event frequency) varies exponentially with age,

$$\lambda(y) = \exp(a + by),$$

where $\lambda(y)$ is the event frequency (mean number of events per year) for a plant of age y, and a and b are parameters to be estimated. This is also called a *loglinear model.* The SAS (1993) program GENMOD found the maximum likelihood estimates of the two parameters, and used asymptotic methods to quantify the uncertainty. The fitted model trend parameter (± 2 std. errors) was

$$b = 0.0868 \pm 0.0778 .$$

In particular, the trend parameter b was clearly nonzero, statistically significant with p-value 0.026. The estimated time for the occurrence frequency to double is 8 years. A 90% confidence interval on this doubling time is from 4.6 yrs to 30 yrs.

The fitted model is shown in Figure 1, with a 90% confidence band around $\lambda(y)$. Also shown in Figure 1 are point estimates and 90% confidence intervals based on separately analyzing the data from each 5-year bin. The point estimates are the maximum likelihood estimates, (observed number of leak events)/(corresponding reactor-years). They are shown as dots, and the confidence intervals are shown as vertical lines. The data are consistent with the exponential modeling assumption (that is, a goodness-of-fit test shows no problems); the figure confirms this, because all the confidence intervals overlap the fitted trend.

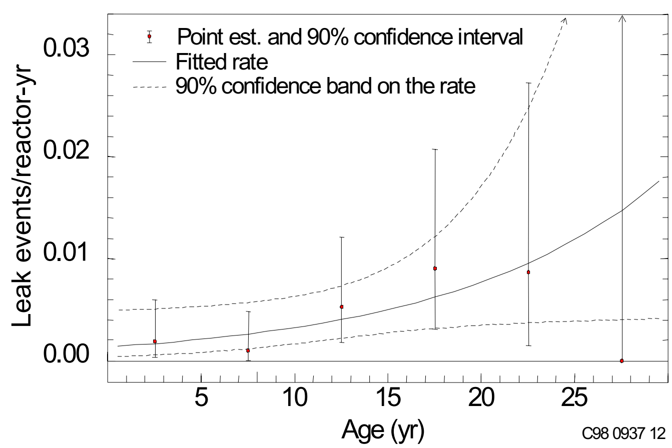

Figure 1. Fitted exponential trend for frequency of thermal-fatigue leak events, with a 90% confidence band on the trend. Also shown are point estimates and 90% confidence intervals for the individual 5-year bins. The trend is statistically significant.

Because the data set is sparse, and the conclusions are potentially important, three other kinds of models were also investigated, as described next.

2.2 *Adjustments in the bin size*

The analysis was repeated with 1-year bins, 10-year bins, and two bins divided at age 8.5 to make the number of reactor-years approximately equal in the two bins. Use of only two large bins lost information, and did not quite show a statistically significant trend. Use of 10-year bins and 1-year bins produced similar results to those for 5-year bins.

This illustrates the following general principles concerning binning of count data. Collapsing the data into few bins loses information, and so is not as powerful for detecting a trend. On the other hand, the analysis uses asymptotic methods, and so is not very accurate when many of the counts are zero. The problem with small counts is most severe when evaluating goodness of fit. No problem was seen with the goodness of fit for this data set, even when the bins corresponded to single years. However, the goodness-of-fit calculations are not necessarily accurate for bins that small. Therefore, the model chosen for this paper uses 5-year bins, with an average of over two events per bin.

2.3 *Methods based on leak dates rather than event counts*

Methods were also used based on the individual ages of the reactors when the leaks occurred. (Atwood 1992 describes the methods and software.) That is, the fundamental data were ages, not the event counts in bins. In the previous model, an increasing leak frequency was revealed by relatively many leak events in old plants and relatively few in new plants. In the present class of models, an increasing leak frequency is revealed by a tendency of leak events to occur late in the observation period rather than early. For plants that had leaks, Figure 2 shows the observation periods by horizontal lines and the leak events by dots. Note that most of the leak events are in the right half of the corresponding observation periods.

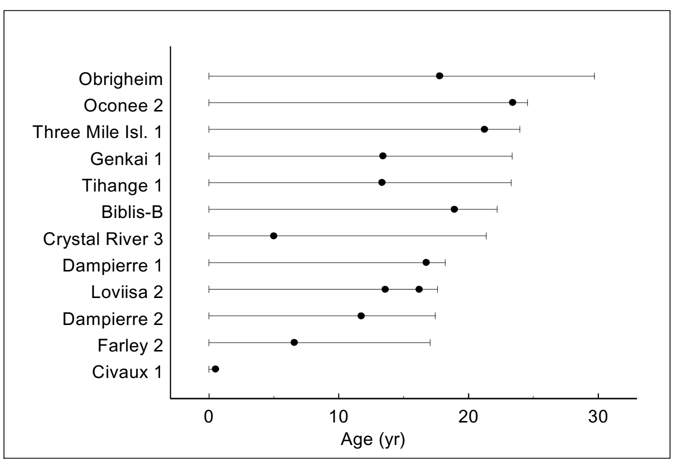

Figure 2. Observation periods and leak events.

2.3.1 *Exponential trend*

First, an exponential trend model was assumed, as in Section 2.1. Two possibilities were considered, one in which the initial leak frequency at age 0 was not necessarily the same at all plants, and one in which the initial leak frequency was forced to be the same at all plants.

The model that allows plants to have different initial leak frequencies yields a statistically significant trend (2-sided p-value = 0.022). The estimated doubling time is 5.8 years, with a 90% confidence interval of from 3.3 years to 42 years. This model can only use the data from the twelve plants where the thirteen leaks have occurred.

A test for equality of the initial leak frequencies did not find a statistically significant difference (p-value = 0.11). Therefore, the model was considered with all the plants assumed to have the same initial leak frequency and same doubling time. This model allowed the data from all 217 plants to be used. The estimated doubling time is 7.6 years, and a 90% confidence interval is from 4.4 to 26 years. These answers are not inconsistent with those found above, in the sense that confidence intervals in Section 2.1 and in this section overlap substantially.

2.3.2 *Power-law trend*

A second type of model was also considered, in which the leak frequency had the form

$$\lambda(y) = a \cdot y^b \tag{1}$$

where y is the plant age in years. Increasing trends correspond to positive values of b. This is a *power-law* or *Weibull* function. When the plants were compared to see if they had the same value of a, Civaux 1 was an outlier. Therefore that plant was dropped from the data set for this analysis. Because the leak event at Civaux 1 occurred nearly at the end of its observation period, the effect of dropping Civaux 1 was conservative — it slightly reduced the estimate of b, and slightly reduced the statistical significance of the trend. Even with Civaux 1 dropped, the parameter b was greater than zero, statistically significant with a p-value of 0.024. The maximum likelihood estimate of b was 1.32, and a 90% confidence interval was (0.22, 2.42). The resulting frequency is shown in Figure 3.

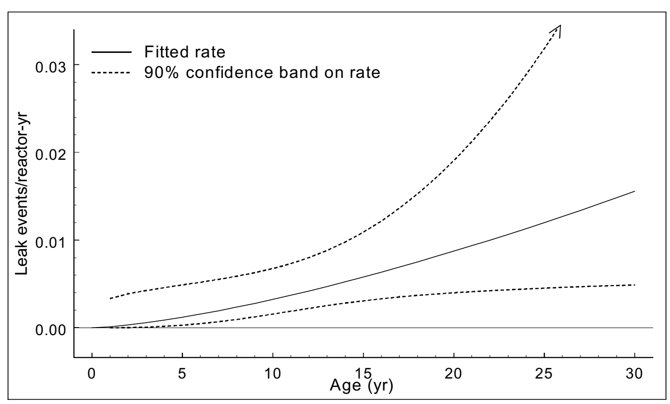

Figure 3. Fitted power-law trend for frequency of thermal-fatigue leak events, with a 90% confidence band on the trend. Civaux 1 is dropped from the data set, reducing the slope of the fitted line slightly. The trend is statistically significant.

This model does not have a single doubling time. Instead, algebraic manipulation of Equation (1) shows that the increase is governed by the rule

$$\lambda(2^{1/b} y) = 2\lambda(y) .$$

Thus the leak frequency is estimated to double from its present value when the plant becomes $2^{1/1.32} = 1.7$ times as old as it is now. For an 11-year old plant, this is similar to the result in Section 2.1. However, for older plants the frequency does not increase as rapidly as under the exponential trend model.

2.4 *Trend in calendar time rather than age*

In many cases, it is difficult to distinguish between a trend with age and a trend in calendar time. Therefore, we considered a model in which the event rate was a function of calendar time rather than of plant age. The form of the model was the same as in Section 2.1. When the model parameters were estimated, the trend was statistically less significant (p-value = 0.050). Moreover, calendar time does not correspond directly to a mechanism, cumulative thermal fatigue. Therefore, we continued to believe that the trend is with age, and any trend in calendar time is indirect, a consequence of the trend in age.

Ideally, one would construct a model that included effects of both age and calendar time, and see which model parameters were statistically significant. However, the data set is much too small to permit this. It is difficult to say how large a data set would be required, but roughly, the number of events should not be much smaller than the number of bins, where each bin defines both a range of calendar time and a range of ages.

In summary, although the data set is sparse, various methods show that the leak frequency is increasing with plant age. Thus, the sparseness of the data does not seem to invalidate the conclusion.

3 CONSEQUENCES AND SAFETY SIGNIFICANCE

3.1 *Interpretation of the statistical analysis*

The above analyses show that, until now, through-wall cracks from thermal fatigue have become more frequent with plant age. The rate of increase is non-zero, but the exact value is quite uncertain.

It is normally unwise to extrapolate conclusions of a model beyond the range of the observed data, for three reasons. (1) Because of parameter uncertainty, the uncertainty on the frequency becomes very large beyond the range of the data. (2) Moreover, the model itself is uncertain. In the present case, an exponential-trend model fits the data well. However, except for one outlying plant, a power-law model also fit the data well, and many other models could be constructed that also would be consistent with the data. Most of the data come from plants in their first 20 years of life (85% of the leaks and 92% of the reactor-years). Therefore, one should be cautious in extrapolating the results of the above analysis to plants older than about 25 years. (3) The above are standard statistical reasons for not extrapolating analysis results. In addition, conditions are changing as some plants take steps to reduce thermal-fatigue cracking. This is a third reason for not extrapolating.

3.2 *Safety significance*

The cracking of PWR primary coolant piping is safety-significant because it reduces the barriers to the release of the radioactive fission products generated during the operation of a PWR, and because a through-wall crack could be a precursor to a loss-of-coolant accident (LOCA).

The reduction of barriers is clear. The precursor argument is outlined here, and discussed more in Poloski et al. (1998). The method follows Beliczey and Schulz (1990) who model the conditional probability of a rupture given a leak as a function of the pipe size. If it is accepted that ruptures follow from leaks with some fixed conditional probability, the statistically valid increasing frequency of through-wall cracks (leaks) implies an increase in the frequency of ruptures, and of LOCAs.

This argument assumes that no intervention takes place. In fact, if one small-break LOCA (SBLOCA) occurs, both the industry and regulators would take significant action to prevent a second, similar one from ever occurring.

One must also realize that even the occurrence of one SBLOCA would not radically change the frequencies used in PRAs. For example, Poloski et al. (1999) give an uncertainty distribution for the SBLOCA frequency, based on no occurrences in many U.S. reactor years. The mean is somewhat smaller than the means in WASH-1400 (US NRC 1975) and NUREG-1150 (US NRC 1990). However, if one SBLOCA had occurred, the recalculated mean would still be slightly smaller than those two means.

Nevertheless, the increasing frequency of thermal-fatigue through-wall cracks is troubling. Therefore, consider some of the issues involved in trying to prevent and detect through-wall cracks.

3.2.1 *Phenomena causing thermal-fatigue cracking and growth*

A number of phenomena have been identified that can cause or contribute to thermal-fatigue cracking.

Turbulent penetration and thermal cycling. Turbulent flow in a main coolant pipe can cause penetration of hot water into a connecting pipe with water of a lower temperature. This introduces the potential for a cycling of temperatures in a portion of the connecting pipe, and thermal fatigue. Depending on the geometry of the piping, the penetration can lead to thermal stratification.

Turbulent mixing. Turbulent mixing of hot and cold coolant can induce local cyclic stresses on the adjacent piping wall. The stress is greatest on the inside surface, and the orientation of any resulting cracks is random.

Thermal shock. Such one-time events can be crack initiators, but probably do not cause crack growth.

Thermal stratification. A horizontal pipe may have hot water at the top and cold water at the bottom. This introduces axial and circumferential bending stresses, with maximum axial stresses near the mixing layer. Intermediate supports will modify the stress distribution. A periodic rising and falling of the mixing layer induces through-wall cyclic stresses that contribute to both crack initiation and growth.

Thermal striping. This results when a pipe has thermal stratification, with a large difference in the flow velocities of the two layers.

Flow-induced vibration. This can be a contributor to thermal fatigue, because the protective sleeve inside a nozzle can be damaged by vibration. If the sleeve breaks loose and moves through the piping, the nozzle is left unprotected, and is subject to thermal fatigue.

All these phenomena have played some role in causing through-wall cracking, but the turbulent penetration and thermal cycling phenomenon has often played the major role. It has caused cracking in both base metal and welds. This phenomenon is understood qualitatively but not well enough for accurate quantitative prediction. In other words, the capability to predict the piping location where this mechanism might produce through-wall cracking is not yet fully developed.

Because of the lack of a reliable predictive capability, monitoring of coolant temperature in the susceptible piping is necessary to identify the piping locations where fatigue cracking might take place. These locations may be included in the in-service inspection program.

The thermal-hydraulic phenomena affecting crack growth are also not well understood. Past experience with thermal-fatigue cracking indicates that the crack growth is slow, and it leads to leakage but does not challenge the structural integrity of the pipe. Experience with a rapidly growing fatigue crack, however, is limited. Such a rapid crack growth occurred at Dampierre 1 in 1997. A portion of the safety injection line was replaced during the repair for the 1996 leakage event (see Table 1). A crack initiated and propagated to 67% through-wall depth in the replaced piping within 8 months after the replacement. This result contradicted the fatigue analysis results for the replaced piping, which indicated that the crack should not initiate for years, even when taking into account local thermal loads revealed by temperature monitoring of the piping (Merle 1998).

Lack of complete understanding of the causes of thermal-fatigue cracking and growth makes it difficult to prevent such cracks by appropriate modifications of the plant. Therefore, consider the issue of detection.

3.2.2 *Detection*

Because of the increasing trend of leak-event frequency, the PWR primary-coolant piping components may fall into a group of high safety-significant components as plants become older. The field experience of thermal-fatigue cracking has revealed several sites that were originally not designed for the observed thermal-fatigue loads. Some of those sites were not included in the in-service inspection. In addition, some cracks have been found with unexpectedly rapid growth. Therefore, to detect fatigue cracking before it becomes a through-wall crack, it is necessary to increase both the inspection frequency and the number of susceptible locations inspected during each inspection. The susceptible locations should include both weld sites and base metal (away from welds), both of which have experienced thermal-fatigue cracking and resulting leakage. In some countries, such inspection of base metal is required.

The inspection itself can be difficult. Current in-service inspection techniques and requirements for branch lines, that is, small diameter piping, have several limitations. ASME Section XI requirements for Class 1 piping with smaller than a 102-mm (4-in.) diameter are not adequate to detect thermal-fatigue damage. The requirements include only surface examination of the welds (ASME 1989), but volumetric examinations are needed to detect thermal-fatigue cracks, which initiate on the inside surface.

It is difficult to detect thermal-fatigue cracks at weld and base-metal sites during in-service inspection when the plant is shut down, especially in small-diameter piping. For example, during in-service inspection in France, many cracks with 33% to 66% penetration were not detected (Merle 1998). It is even more difficult to size these cracks. Qualified in-service inspection techniques are needed to characterize thermal-fatigue cracks in PWR branch lines.

At present, inspection of susceptible base-metal sites (away from welds) is generally not required at U.S. PWRs. But, as mentioned above, thermal-fatigue leaks can occur in base metal. Recently, because of the 1996 leak event at Dampierre 1, the in-service inspection program for the French PWRs has been revised to include the inspection of base metal of the unisolable portion of the safety injection lines (Gauthier 1998). Shah and Ware (1994) also recommended inspection of base-metal sites susceptible to thermal-fatigue cracking. Risk-informed inspections may address this inspection need at U.S. PWRs.

4 SUMMARY

We have analyzed worldwide data for PWR primary coolant leak events caused by thermal fatigue. Various statistical analyses all reach the same conclusion, that the frequency of such events is increasing with plant age. However, the estimated model parameters have large uncertainty. Also, different models (different equations for the frequency as a function of time) can be used. For these reasons, even if no plants were working to reduce the effect of thermal fatigue, one could not justify a quantitative extrapolation of the trend to plants much older than 25 years.

The following are the difficulties in arresting this increasing frequency.

- The phenomena causing thermal fatigue are not understood well enough to give quantitative prediction of the locations of through-wall cracks. To identify these locations temperature monitoring is needed.
- Based on this monitoring, locations that are susceptible to thermal fatigue may be inspected, both welds and base metal. However, qualified techniques are needed to detect thermal-fatigue cracks.
- Most thermal-fatigue cracks grow slowly, but some have grown quite rapidly. Experience with fast-growing cracks is very limited. The frequency of inspection should take into account this limited knowledge of the growth rate.

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